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Non-Destructive Examination for Ageing Management of Indian Nuclear Power Plants

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Abstract

In India there is a comprehensive ageing management programme for all its nuclear power plants. Non-destructive examination is a vital link in this regard, as it provides valuable inputs in the form of flaw characteristics and degradation of material properties for fitness-for-service assessment of critical components in nuclear facilities. This paper highlights NDE methodologies as being used and developed for in-service inspection of critical components of Indian Boiling Water Reactors and Pressurized Heavy Water Reactors.

Keywords: BWR, PHWR, Ultrasonic Testing, IGSCC, Hydrogen Damage

1.0 Introduction

The twin boiling water reactors (BWRs) at Tarapur were the first nuclear power plants to be set-up in India. Subsequently, the three stage Indian Nuclear Power Programme commenced with the setting up of pressurized heavy water reactors (PHWRs). India currently has 16 operating PHWRs and few more under construction. There is a comprehensive ageing management programme in place for all the operating nuclear power plants. Periodic in-service inspection of critical components in these reactors goes a long way in assuring their structural integrity thereby guaranteeing the safe operation of nuclear plants. Non-Destructive Examination (NDE) techniques play a crucial role in this regard. The BWRs in India are in operation for more than three decades. The in-service inspection programme for BWR components is based on the guidelines of ASME B&PV Code Sex. XI. Some of the critical components which are periodically examined during re-fuelling outages include: primary pipelines, feed water nozzles, turbine blades, steam generators, core internals, pressure vessel and its internals, etc. Intergranular stress corrosion cracking is the generic problem in these reactors. The components prone to such attack are periodically monitored by NDE for initiation and growth of IGSCC. In Indian PHWRs, zirconium alloys is predominantly used as the material of construction for core components. The integrity of pressure tube, which carries the fuel and the high temperature high pressure coolant, is central to the safety of PHWRs. The pressure tube, which is prone to hydrogen attack viz. delayed hydride cracking, hydride blistering and hydride embrittlement, is periodically examined during in-service inspection. This paper highlights NDE methodologies as being used and developed for in-service inspection of critical components of Indian BWRs and PHWRs and their role in ageing management of these components.

2.0 Monitoring Initiation and Growth of Intergranular Stress Corrosion Cracking in Boiling Water Reactors

One of the significant failures reported in 1990s in BWRs all over the world, is that of reactor core shroud cracking. The cracks were confined to weld heat affected zones and the mechanism of cracking was identified as intergranular stress corrosion cracking (IGSCC) and irradiation assisted stress corrosion cracking (IASCC). Core Shroud of BWR at Tarapur, India, is 25 mm thick AISI 304 austenitic stainless steel cylinder. It partitions the feed water in the reactor vessel downcomer annulus region from the coolant water flowing upwards through the reactor core. It provides structural support to the core and maintains its geometry. The core shroud has nine circumferential welds out of which the top four welds, H1, H3, H4A and H4B, are accessible for inspection from inside. While H3 is in the form of 12 lug welds, H4A and H4B are circumferential welds. These four welds are inspected by visual and ultrasonic examination. While the visual examination is carried out using an underwater radiation resistant camera, special probe holders and manipulators are used for ultrasonic examination.

The core shroud inspection is being carried out during every re-fuelling outage since 1995. No crack-like indication has been observed in any of the welds examined [1]. The effectiveness of visual examination for detection of fine cracks was qualified on SRCS (Sensitivity, Resolution, Contrast Standard) Panels. These panels comprise of stainless steel plate to which four wires of varying diameters (15 to 70 microns) are attached. These panels are kept underwater and the wires are viewed by a radiation resistant underwater camera kept at a distance applicable during actual in-service inspection. These panels were fabricated in the laboratory and used for on-site qualification.

For ultrasonic examination, the technique for IGSCC detection is based on angle beam examination of the heat affected zone. For circumferential welds H4A and H4B, the probe holder comprises of three ultrasonic transducers: two angle beam and one normal beam. One of the angle beam transducers is used for the top HAZ and the other for the bottom HAZ. The normal beam transducer is used to pick-up the vessel ID signal which is used as a reference for ensuring the radial beam direction during angle beam examination. The probe holder is housed in a specially designed mechanism called as Carriage for Advancing and Retracting Transducer (CART). The CART assembly (Figure 1) was designed and fabricated in collaboration with Nuclear Power Corporation of India Limited. The objective if CART is to achieve an extended coverage (instead of spot checking) during ultrasonic examination of H4A circumferential weld in the core shroud. The CART is connected to the Grapple operated manipulator with the help of which it can be taken to the desired azimuth location of the H4A weld. The development of CART system helped to examine the larger length of weld joint in a very short time.

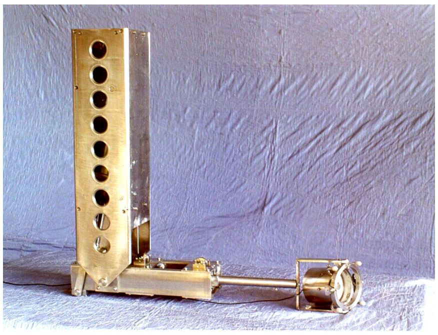


Figure 1: CART assembly

Since the bottom five welds are not accessible for any examination, extensive stress analysis for safety assessment of the cracked shroud (cracks assumed at various weld locations) has been carried out at normal operating loads and loads due to Re-circulation Line Break (RLB), Main Steam Line Break (MSLB) and Seismic events [2]. The results indicate that the safety functions like control rod movement, core spray and poison injection are not impaired under these abnormal conditions.

IGSCC is also a generic problem in primary pipelines of BWRs. These are examined periodically as per the guidelines of ASME Boiler and Pressure Vessel Code Sec. XI. The code calls for 100% ultrasonic examination of the welds and the heat affected zones by using angle beam shear wave technique and 10% wall thickness deep machined notch as reference defect standard. Over the years, many IGSCC failures have been observed in these pipelines and corrective action has been taken. The old pipelines have also been replaced with the new pipelines of IGSCC resistant material. The new pipelines are inspected after installation and also periodically during re-fuelling outage for monitoring IGSCC.

The limitation of conventional ultrasonic technique based on amplitude comparison for evaluation of IGSCC depth is well known. The difficulty arises due to the poor reflectivity of IGSCC to ultrasonic waves. In order to overcome these limitations, two approaches were standardized. The first approach uses a known depth IGSCC as reference defect standard instead of machined notch. This approach takes care of the difference in reflectivity of IGSCC in the component and the reference standard. The second approach employs tip-diffraction techniques for sizing IGSCC. In these techniques, sizing is based on monitoring the time of travel of the reflected/diffracted signals from the crack extremists.

In order to standardize the above two approaches, IGSCC was generated in 25 mm thick stainless steel plates. The plate containing a deep, circular groove was first sensitized at 677 C for 1 hour. Subsequently, weld was deposited in the groove under restraint. This ensured residual stresses of significant magnitude in the heat affected zone. The plate was then exposed to polyphonic acid that is known to generate IGSCC sensitized austenitic steel at room temperature. After overnight exposure, IGSCC was generated in the plate (Figure 2). The intergranular nature of crack was confirmed by in-situ metallography.

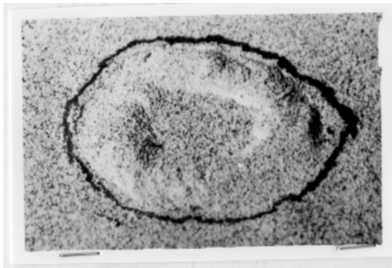


Figure 2: IGSCC generation in laboratory

The plate containing IGSCC was then subjected to ultrasonic examination. Three different techniques are used for sizing viz. (i) using 10% wall thickness deep notch as reference defect standard (as per ASME B&PV Code Sec. XI guidelines), (ii) using a known depth IGSCC as reference defect standard and (iii) tip-diffraction technique. The results of this study are shown in Table 1:

Table 1: IGSCC sizing by different ultrasonic techniques

Sr. No.	Depth of IGSCC measured by metallography (mm)	Depth of IGSCC estimated using 10% deep notch as reference standard (mm)	Depth of IGSCC using known depth IGSCC as reference standard (mm)	Depth of IGSCC estimated using tip diffraction technique (mm)
1	7.7	2.4	7.7 *	7.7
2	6.7	2.1	4.8	6.6
3	7.0	1.5	3.0	7.0
4	4.3	1.9	3.2	4.9
5	5.8	1.2	2.4	5.6

** Known depth, used for reference*

The results of the investigation indicate that, the depth sizing accuracy for IGSCC is extremely poor using notch as reference defect standard. The sizing accuracy is improved by using known depth IGSCC as reference standard, but the best accuracy is obtained by tip-diffraction technique. [3,4].

3.0 In-service inspection of Coolant Channels in Pressurized Heavy Water Reactors

The pressurized heavy water reactors (PHWRs) are the mainstay of Indian Nuclear Power Programme. Currently there are 16 operating PHWRs and many more under construction. The Indian PHWR [8] consists of few hundreds coolant channels. A coolant channel assembly comprises of a horizontal pressure tube (PT) carrying fuel and hot coolant, enclosed by a concentric calandria tube (CT). These tubes are separated by four garter spring spacers, which prevent hot PT to come in contact with cold CT (Fig. 3). There are 306 coolant channels in a typical 220 MWe and 380 channels in 540 MWe Indian PHWR. The operating

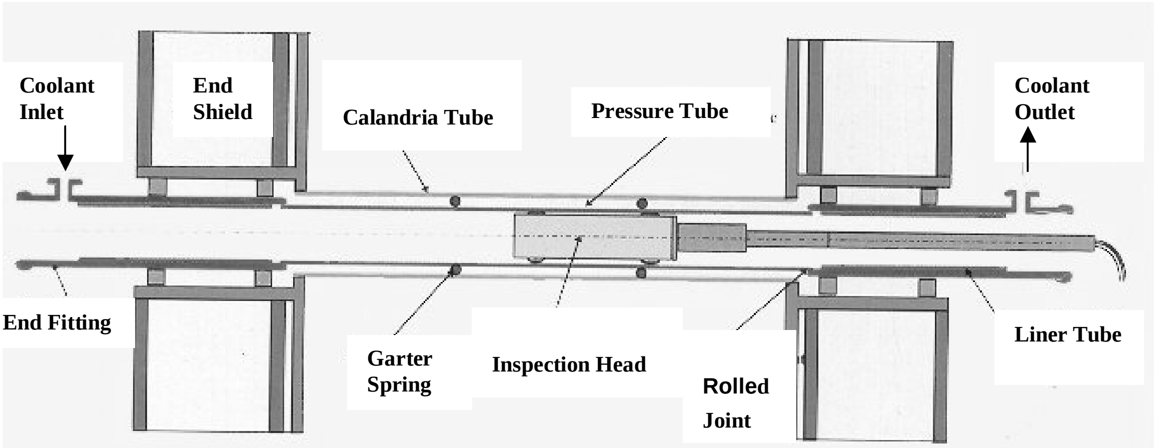


Figure 3: Typical PHWR coolant channel assembly

conditions in PHWR lead to the degradation in pressure tube with respect to (i) dimensional change (creep and growth), (ii) deterioration in mechanical properties (hardening and embrittlement) thereby reducing its flaw tolerance, (iii) the growth of existing flaws, which were too small or ‘in-significant’ at the time of installation, and (iv) initiation and growth of new flaws. The pressure tube, which is made of zirconium alloy (Zircaloy – 2 or Zr - 2.5% Nb), undergoes corrosion in aqueous environment during service. This reaction releases hydrogen, a part of which gets absorbed in the pressure tube material. The absorbed hydrogen is responsible for the two most commonly observed degradation mechanisms, which can limit the life of a pressure tube viz. delayed hydride cracking and hydride blister formation and cracking. In PHWRs with zircaloy -2 pressure tubes, an additional mechanism in the form of accelerated oxidation and hydriding can lead to its premature retirement from service, when the oxide layer thickness on its ID surface exceeds the threshold limit. The integrity of pressure tubes is central to the safety of PHWRs. In order to ensure this at all times during its service, the pressure tubes are periodically examined by non-destructive examination (NDE) techniques. This inspection provides valuable inputs on the presence or absence of flaw in pressure tube and its characteristics to designers, plant operators and regulatory authorities for fitness-for-service assessment.

One of the most likely mechanisms of crack initiation and propagation in PHWR pressure tube is Delayed Hydride Cracking (DHC). DHC initiation is characterized by threshold stress intensity K_{IH} ($K_{IH} \approx 10$ MPa$\sqrt{m}$ for zircaloy- 2) and concentration of hydrogen. The critical flaw size that is required to initiate DHC is given by [9],

$$a = 0.4\,(K_{IH} / \sigma)^2 \tag{2}$$

where, σ is the maximum tensile stress. The objective of in-service inspection is to detect the flaw of this order and even smaller. DHC would lead to leak-before-break if the crack length at leakage is less than the critical crack length. DHC is more likely to initiate in the roll-joint area due to the presence of high residual stresses. Although, large number of Zr-

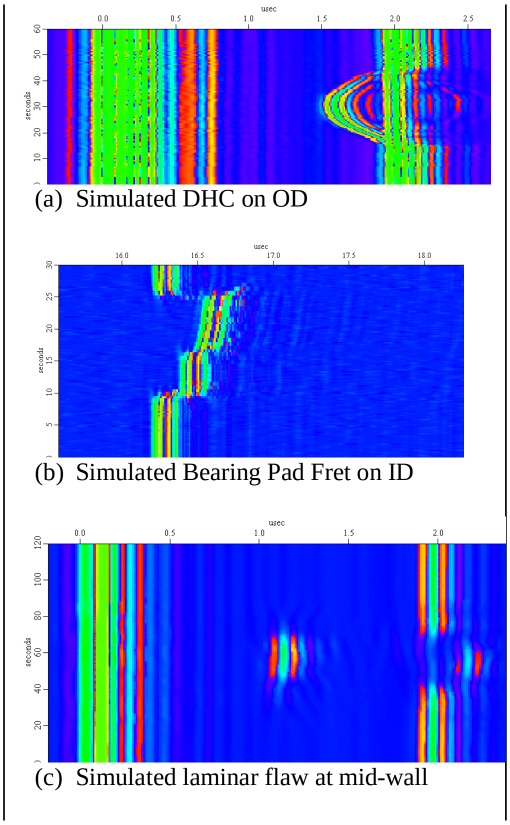
2.5% Nb pressure tubes has shown failure in this region due to DHC, no such failure has been observed in zircaloy-2 pressure tubes [10].

Another hydrogen related failure mechanism in pressure tube is blister cracking. During reactor operation, pressure tube sags due to axial creep and may come in contact with calandria tube, if the garter springs are shifted from their design locations. This contact results in formation of cold spot on the outside surface of the pressure tube. Under the influence of temperature gradient, hydrogen from the other locations in the pressure tube migrates to the cold spot. Due to the limited solubility of hydrogen in zircaloy, this accumulation of hydrogen at the cold spot results in formation of brittle zirconium hydride blister. The blister cracks after reaching a critical size. Further crack propagation can be by DHC. For zircaloy-2 pressure tube, the critical blister size is of the order of 0.65 mm [11]. Presence of cracked blister is a matter of concern for safety of pressure tubes. It is therefore important to detect the presence of un-cracked blister non-destructively, so that corrective action can be taken much before the blister reaches the critical size. Several models on growth of hydride blisters are being developed [12].

In India, in-service inspection of PHWR coolant channels is carried out using a semi-automated channel inspection tool known as BARCIS (BARC channel inspection system). BARCIS consists of inspection head (which carries ultrasonic and eddy current sensors), special sealing plug and drive mechanism. The NDE capabilities in BARCIS include ultrasonic wall thickness measurement and ultrasonic flaw detection in pressure tube, eddy current detection of garter spring location and tilt, eddy current estimation of annular gap between pressure tube and calandria tube and eddy current flaw detection in pressure tube on inside surface. During in-service inspection of coolant channel, the inspection head is moved from inside of the pressure tube in a systematic manner from one end to the other. Indications, if any, are recorded and evaluated for acceptance/rejection [13].

The authors' laboratory represented India during International Atomic Energy Coordinated Research Programme (CRP) on 'Intercomparison of Techniques for Pressure Tube Inspection and Diagnostics'. A total of seven laboratories from six countries participated in this CRP. The primary objective of this CRP was to arrive at the most effective NDE techniques for pressure tube inspection. The CRP involved 'blind tests' on pressure tube samples containing artificial flaws that resemble real flaws of concern such as delayed hydride cracking, fretting damage on pressure tube ID due to bearing pad or debris, lap-type defect and laminar flaws. The inter-comparison of NDE techniques based on the results of investigation of pressure tube samples highlights the most reliable and accurate NDE method (ultrasonic, eddy-current or a combination of both) and also a specific technique for that NDE method (time-of-flight monitoring, amplitude monitoring, C-scan image, etc.) for detection and characterisation of various kinds of flaws encountered in pressure tubes. This information is useful for the heavy water reactor community to improve the tools being used for pressure tube inspection and diagnostics, by modifying the existing techniques or adapting new ones, so that the inspection is carried out in the most effective manner [14, 15]. Indian team detected all the flaws in all the pressure tube samples. These flaws were very accurately characterized by advanced ultrasonic techniques.

As part of this CRP, seven pressure tube samples were examined by ultrasonic testing. The flaws in these samples were characterized by B-scan and C-scan imaging. The images provided useful information on the nature of flaws such as DHC, debris fret, bearing pad fret, lap-type flaw, etc. Figure 4 shows the B-scan images for some of the flaws in pressure tube samples. Figure 4a shows the B-scan image for the DHC on OD (simulated by a fine notch). The image is collected by circumferential movement of a normal beam OD transducer. X-axis represents the time-of-flight and the Y-axis represents the transducer travel. The signal on the left is from pressure tube ID and the one on the right is from pressure tube OD. The semi-elliptical profile of the notch, which is typical of DHC, is clearly seen in the image. Figure 4b shows the B-scan image for a bearing pad fret (simulated by ID groove). The image shows the change in surface profile of the ID signal at the flaw location. This image is collected by moving the ID focused normal beam transducer along the width of the flaw. Figure 4c shows the B-scan image for a laminar flaw, which is simulated by a flat bottom reflector. The flaw is at the mid-wall of the pressure tube. The image shows the signals from pressure tube ID and OD and an additional signal from the laminar flaw close to the centre of the pressure tube thickness.



(a) Simulated DHC on OD

(b) Simulated Bearing Pad Fret on ID

(c) Simulated laminar flaw at mid-wall

Figure 7: B-scan images pf simulated flaws in pressure tube

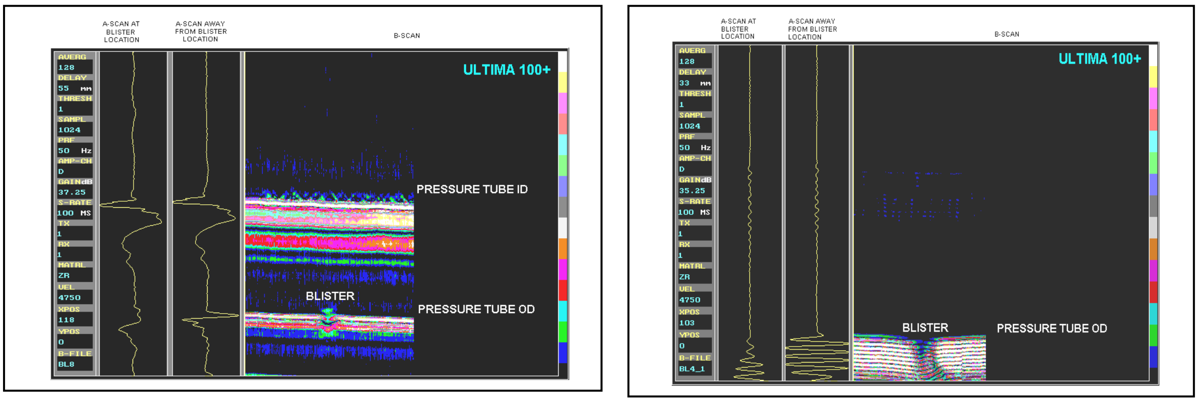
Detection of un-cracked hydride blister in pressure tube poses several challenges, as they cannot be ‘sensed’ by conventional ultrasonic techniques. There is not much difference in the acoustic impedance of zircaloy and zirconium hydride. As a result, the interface between the blister and the parent metal does not reflect any significant amount of incident ultrasonic energy. As a result, new approaches are standardized for reliable detection of hydride blisters. Ultrasonic examination techniques based on ultrasonic velocity ratio measurement, ultrasonic B-scan and C-scan imaging, etc are standardized for this purpose [16]. These techniques are based on difference in ultrasonic velocity in zircaloy and zirconium hydride. Table 2 shows the difference in sound velocity in zircaloy and zirconium hydride:

Table 2: Sound velocity in zircaloy-2 and zirconium hydride

Material	Longitudinal Velocity (V_L) m/sec	Shear Velocity (V_T) m/sec	Velocity Ratio (V_L/V_T)
Zircaloy-2	4750	2350	2.02
Zr-Hydride	5400	1900	2.84

For in-situ applications, the actual thickness of the pressure rube is not known. As a result the actual sound velocity cannot be measured. Velocity ratio is a useful parameter for such applications.

Ultrasonic imaging was used as a tool for detection of hydride blisters in pressure tube. A pressure tube sample containing laboratory grown blisters was scanned in an immersion condition using normal beam longitudinal wave and angle beam shear wave. B-scan images are collected during the travel. These images are shown in Figure 5a & 5b. Figure 5a shows the B-scan image using longitudinal wave. Since the velocity of longitudinal wave is higher in Zr-hydride as compared to zircaloy, the time of travel fro the backwall echo is less at the blister location. This is clearly seen in the image. While using the shear wave (Figure 5b), the reverse happens. This is because of the lower shear wave velocity in Zr-hydride as compared to ziracaloy -2.



a) Longitudinal wave b) Shear wave

Figure 5: B-scan images for hydride blister in zircaloy pressure tube

4.0 Conclusion

The objective of NDE during periodic in-service inspection of nuclear reactor components is to detect flaws, characterize them and ascertain degradation in material properties. The above case studies demonstrate the importance of NDE in (i) ensuring that no component in nuclear plants pose an unacceptable integrity risk prior to the next scheduled inspection and (ii) better understanding of degradation mechanisms and the rate at which they occur. This information is vital for effective ageing management of these components towards assurance of their structural integrity during operation, estimation of residual life and their life extension. Developmental activities have been initiated to apply advanced NDE methods for in-service inspection of critical nuclear power plant components. Some of these areas include: (i) ultrasonic phased array inspection for pipeline welds, core shroud welds, pressure vessel welds, feed water nozzles, turbine blades, etc, (ii) guided wave inspection for monitoring corrosion damage in pipelines and feeders, (iii) multi-NDE approach using eddy current, ultrasonics, magnetic flux leakage for heat exchanger tubes, (iv) non-destructive characterization of material property degradation due to hydrogen and irradiation, etc. In future the major thrust will be on further improvement in flaw characterization capabilities, monitor material property degradation and bring those components within the fold of in-service of inspection, which are currently in-accessible. The new generation NDE techniques will play a vital role in this regard. There will also be major emphasis on modeling and simulation, robotics, sensor development, data fusion, etc. to overcome the challenges faced during ageing management of critical components in Indian nuclear plants.

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